Competency 1.3 **Radiation protection personnel shall demonstrate a working level knowledge of principles and concepts for internal and external dosimetry.**

1. SUPPORTING KNOWLEDGE AND/OR SKILLS

a. Define the following terms:
   - Dose equivalent
   - Shallow dose equivalent
   - Deep dose equivalent
   - Effective dose equivalent
   - Committed dose equivalent
   - Committed effective dose equivalent
   - Total effective dose equivalent
   - Whole body
   - Extremity
   - Lens of the eye dose equivalent
   - Derived air concentrations (DAC)
   - Annual limit on intake (ALI)
   - Quality factor
   - Weighting factor
   - Roentgen
   - Rad
   - Rem
   - Sievert
   - Gray
   - Stochastic effects
   - Nonstochastic (deterministic) effects

b. Describe the various types of bioassays, their applications and limitations.

c. Discuss the methods of reducing dose from internally deposited radionuclides.

d. Discuss the process used to evaluate dose based on bioassay results.

e. Describe the principle of operation, proper use, placement, function, and type of radiation detected by the following dose-measuring instruments:
   - Thermoluminescent dosimeter, including Albedo dosimeter
   - Pocket dosimeter (quartz fiber and electronic)
   - Film badge
   - Personnel nuclear accident dosimeter

f. Discuss the concepts of International Commission on Radiological Protection (ICRP) Publications 26 and 30 as they relate to internal and external dosimetry.

g. Discuss various methods used to estimate worker exposure in the absence of individual monitoring results.

h. Given airborne radioactivity concentration, DAC value, and worker occupancy time; evaluate resulting worker dose.

2. SUMMARY

Definitions

dose equivalent (H): The product of the absorbed dose (D) (in rad or gray [Gy]) in tissue, a quality factor (Q), and all other modifying factors (N). Dose equivalent is expressed in units of rem (or sievert [Sv]) (1 rem = 0.01 Sv).

shallow dose equivalent: The dose equivalent deriving from external radiation at a depth of 0.007 cm in tissue.

deep dose equivalent: The dose equivalent derived from external radiation at a tissue depth of 1 centimeter in tissue.

effective dose equivalent (H_E): The summation of the products of the dose equivalent received by specified tissues of the body (H_T) and the appropriate weighting factors (W_T); i.e., $H_E = \Sigma W_T H_T$. It includes the dose from radiation sources internal and/or external to the body. The effective dose equivalent is expressed in units of rem (or Sv).

committed dose equivalent ($H_{T,50}$): The dose equivalent calculated to be received by a tissue or organ over a 50-year period after the intake of a radionuclide into the body. It does not include contributions from radiation sources external to the body. Committed dose equivalent is expressed in units of rem (or Sv).

committed effective dose equivalent ($H_{E,50}$): The sum of the committed dose equivalents to various tissues in the body ($H_{T,50}$), each multiplied by the appropriate weighting factor (w_T); i.e., $H_{E,50}=\Sigma w_T H_{T,50}$. Committed effective dose equivalent is expressed in units of rem (or Sv).

total effective dose equivalent (TEDE): The sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). Deep dose equivalent to the whole body may be used as effective dose equivalent for external exposures.

whole body: For the purposes of external exposure, head, trunk (including male gonads), arms above and including the elbow, or legs above and including the knee.

extremity: Hands and arms below the elbow, or feet and legs below the knee.

lens of the eye dose equivalent: The external exposure of the lens of the eye and is taken as the dose equivalent at a tissue depth of 0.3 cm.

derived air concentration (DAC): For the radionuclides listed in Appendix A of 10 CFR 835, the airborne concentration that equals the ALI divided by the volume of air breathed by an average worker for a working year of 2,000 hours (assuming a breathing volume of 2,400 m^3). For radionuclides listed in Appendix C of 10 CFR 835, the air immersion DACs were calculated for a continuous, nonshielded exposure via immersion in a semiinfinite atmospheric cloud. The values are based upon the derived airborne concentration found in Table 1 of the U. S. Environmental Protection Agency's Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion,*

and Ingestion, published September 1988.

annual limit on intake (ALI): The derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the Reference Man (ICRP Publication 23) that would result in a committed effective dose equivalent of 5 rem (0.05 Sv) or a committed dose equivalent of 50 rem (0.5 Sv) to any individual organ or tissue. ALI values for intake by ingestion and inhalation of selected radionuclides are based on Table 1 of the U.S. Environmental Protection Agency's Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, published September 1988.

quality factor [Q]: The principal modifying factor used to calculate the dose equivalent from the absorbed dose; the absorbed dose (expressed in rad or Gy) is multiplied by the appropriate Q.

Radiation Type	Quality Factor
X-Rays, Gamma Rays, positrons, electrons (including beta particles)	1
Slow Neutron, ≤ 10 keV	3
Fast Neutron, ≥ 10 keV	10
Protons and singly-charged particles of unknown energy with rest mass greater than 1 amu	10
Alpha particles and multiple charged particles (and particles of unknown charge) of unknown energy	20

weighting factor (w_T): The fraction of the overall health risk, resulting from uniform, whole body irradiation, attributable w_T to specific tissue (T). The dose equivalent to the affected tissue, H_T, is multiplied by the appropriate to obtain the H_E contribution from that tissue.

Organ or tissue, T	Weighting Factor, w_T
Gonads	0.25
Breasts	0.15
Red bone marrow	0.12
Lungs	0.12
Organ or tissue, T	**Weighting Factor, w_T**
Thyroid	0.03
Bone surfaces	0.03
Remainder[1]	0.30
Whole body[2]	1.00

[1] "Remainder" means the five other organs or tissues with the highest dose (e.g., liver, kidney, spleen, thymus, adrenal, pancreas, stomach, small intestine, and upper large intestine). The weighting factor for each remaining organ or tissue is 0.06.

[2] For the case of uniform external irradiation of the whole body, a weighting factor (w_T) equal to 1 may be used in determination of the effective dose equivalent.

roentgen (R): A unit of exposure to ionizing radiation. It is that amount of gamma or x-rays required to produce ions carrying one electrostatic unit of electrical charge in one cubic centimeter of dry air under standard conditions. In SI units, one R produces an electrical charge in air of 2.58 E-4 coulombs per kg. Named after Wilhelm Roentgen, a German scientist who discovered x-rays in 1895.

rad: Unit of absorbed dose. One rad is equal to an absorbed dose of 100 ergs per gram or 0.01 joule (J) per kilogram (kg) (0.01 Gy).

rem: Unit of dose equivalent. Dose equivalent in rem is numerically equal to the absorbed dose in rad multiplied by a Q, distribution factor, and any other necessary modifying factor (1 rem = 0.01 Sv).

sievert (Sv): SI unit of any of the quantities expressed as dose equivalent. The dose equivalent in Sv is equal to the absorbed dose in Gy multiplied by the Q (1 Sv = 100 rem).

gray (Gy): SI unit of absorbed dose. One Gy is equal to an absorbed dose of 1 J per kg (100 rad).

stochastic effects: Malignant and hereditary diseases for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose without a threshold for radiation protection purposes.

nonstochastic effects: Effects due to radiation exposure for which the severity varies with the dose and for which a threshold normally exists (e.g., radiation-induced opacities within the lens of the eye).

Bioassays

The objective of bioassay is to detect and/or determine the amount of radioactive material in the body. Bioassays may be routine, performed on a regular basis, or event-driven such as after a suspected intake. Two classifications of bioassays are in vivo, meaning the radioactive material is detected while it is still in the body, and in vitro, meaning a sample is taken out of the body and counted. The following are some common types of bioassays:

- In vivo
  - Whole body count
  - Lung count
- In vitro
  - Urinanalysis
  - Fecal analysis
  - Blood analysis
  - Sputum and nasal smears
  - Sweat
  - Breath analysis

The type of bioassay will be determined by:

- The types(s) of radiation emitted by the radionuclide taken into the body, and the type(s) of radiation emitted by the daughter radionuclides
- Chemical solubility of the radionuclide
  - soluble
  - insoluble

<u>In vivo Bioassay</u>

Whole-body counting provides data for the assessment of gamma and x-ray emitting radionuclides deposited in the body. In some cases beta emitting radionuclides can be detected by the bremsstrahlung radiation. Gamma, x-ray, and bremsstrahlung radiation originating in the body can leave the body and be detected by a variety of radiation detectors, including sodium iodine (NaI); a phoswich detector which is a combination NaI and cesium iodide; and intrinsic germanium (HPGe). These detectors are positioned over the body to detect photons emitted from particular radionuclides. No samples are required to be taken from the body. For more information on these detectors, see Radiation Protection Competency 1.5.

The detectors are often placed to detect radioactivity emitted from the whole body or from a localized portion of the body. Lung counters, for example, are used to detect inhaled radioactive material in the lungs.

They are widely used at DOE facilities to assist in the detection of inhaled uranium (U) and plutonium (Pu).

Whole-body counting is capable of identifying specific isotopes, especially iron-59 (Fe-59), cobalt-60 (Co-60), zinc-65 (Zn-65), ruthenium-106 (Ru-106), iodine-131 (I-131), and cesium-137 (Cs-137).

Whole-body counting procedures have certain limitations in terms of sensitivity and number of radionuclides that can be readily determined. Sensitivity is in part a function of the background and the background count time. The following chart shows some of the capabilities of the Hanford whole-body counter.

Radionuclide	**Count Time (Secs)**	**Body Position**	**Detection Sensitivity (nCi)**
Pu-239	2000	Lung	2.4
Pu-239	3000	Bone	5.4
Am-241	2000	Lung	0.18
I-125	2000	Thyroid	0.004
Cs-137	200	Whole Body	3
Co-60	200	Whole Body	3
K-40	200	Whole Body	15

The major benefit of whole-body counting is obtained after contamination has been internally deposited and it is therefore of little value in mitigation of the radiological effects of the deposition. Further, such techniques of assessment do not reveal the sources of the radiocontaminants or their mode of entry into the body. Whole-body counting is, therefore, only a small part of a total radiological protection program.

In vitro Bioassay

In vitro bioassay is required for nonpenetrating radiations such as alpha and beta. Since the energy of these radiations cannot leave the body, a sample must be taken from the body, (usually body fluids) and analyzed for radioactive material.

The body material used for analysis depends on the chemical and physical form of the nuclide, its mode of entry into the body and how the material is biologically processed in the body. Bioassay procedures are most applicable to nuclides that are soluble because they transfer to the blood and body fluids. The sampling of feces, however, can be used to determine the amount of insoluble isotopes in the body. One advantage of bioassay over whole-body counting is that bioassay can be used to detect alpha and beta emitters. Some isotopes that can be detected by bioassay are tritium, radium, thorium, uranium, plutonium, iodine, cesium,

phosphorus, strontium, barium, and ruthenium.

Bioassay procedures include urinalysis; blood analysis; analysis of sputum, nasal smears, and feces; and breath analysis.

- Urinalysis is used mainly to detect uptake of soluble beta and alpha emitters. Generally tritium (H-3) uptakes are measured by urinalysis.
- Analyses of sputum, nasal smears, and feces are used to detect insoluble beta and alpha emitters Insoluble contaminates are usually difficult when measuring body burdens. If an insoluble radioisotope is ingested, it can pass relatively easily throughthe gastrointestinal tract. In the case of inhalation, the contamination is eventually carried up the respiratory tree, intothe epiglottis, and into the gastrointestinal tract where it is passed outside of the body and canbe detected through a fecal sample. Nasal smears of Pu may indicate an intake.
- Breath analysis is used to determine if radioactive gases are present in the body. The body burden of radium-226 (Ra-226) can be determined bydetecting the amount of radon-222 (Rn-222) exhaled. Also, carbon-14 (C-14) uptakes can be detected as thecarbon is metabolized and exhaled as carbon dioxide CO_2.

Internal Dose Reduction Methods

Respiratory protection should be used to reduce the level of internal dose in airborne radioactivity areas. The respirator should have a protection factor (defined as the amient airborne concentration divided by the concentration inhaled) adequate enough to lower the inhaled radioactivity below the allowable occupational air concentration. Other conditions for using respirators include:

- Training for proper use by the wearer
- Respirator testing for functionality
- Written procedures for selecting, fitting, and maintaining the respirator
- Physician approval before intial fitting of a respirator
- For detailed information on respirator use and regulations see:*Practices for Respiratory Protection*, American National Standards Institute (ANSI Z88.2-1992).

In general, speed is essential when attempting to reduce internal dose by drug administration. For actinide americium (Am) or curium (Cm), experiments have shown that about 75% of the bone deposition is complete in one hour. It is very important to note that any medical administrations must be done under the care of a physician.

Blocking Agents

A blocking agent saturates the metabolic processes in a specific tissue with the stable element and reduces uptake of the radioactive forms of the element. As a rule, these must be administered prior to, or almost

immediately, after the intake for maximum effectiveness and must be in a form that is readily absorbed. The most well known blocking agent is stable iodine given as potassium iodide (KI), which is used to saturate the thyroid gland thus preventing uptake of radioactive iodine in the thyroid. Remember, any administration of blocking agents must be under medical instruction and supervision as adverse side effects are possible.

Diluting Agents

A diluting agent is a compound which includes a stable form of the nuclide of concern. By introducing a large number of stable atoms in the body, the statistical probability of the body incorporating radioactive atoms is reduced. A good example is increasing water intake following H-3 exposure. Diluting agents can also involve the use of different elements which the body processes in the same way. This type of treatment is called displacement therapy. A common form of this is the use of calcium to reduce deposition of strontium (Sr). The compound used must be as readily absorbed and metabolized as the compound that contains the radioisotope.

Mobilizing Agents

A mobilizing agent is a compound that increases the natural turnover process, thus releasing some forms of radioisotopes from body tissues. Mobilizing agents are usually most effective within 2 weeks after exposure; however, use for extended periods may produce less dramatic reductions.

Expectorants and Inhalants

These are used to increase flow of respiratory tract excretions. Thus far, this type of therapy has not been proven successful in removing radioactive particles from all areas of lungs.

Lung Lavage

This involves multiple flushing with appropriate fluid to remove radioactive materials in the lungs. Usually limited to applications where resulting exposures would result in appearance of acute or subacute radiation effects.

Chelating Agents

A chelating agent is a compound which acts on insoluble compounds to form a soluble complex ion which can then be removed through the kidneys. It is commonly used to enhance elimination of transuranics and other metals. The therapy is most effective when begun immediately after exposure if metallic ions are still in circulation, and is less effective once metallic ions are incorporated into cells or deposited in tissue such as bone.

Common chelating agents include ethylenediaminetetraacetic acid (EDTA) and diethylenetriaminepentacetic acid (DTPA).

- Calcium edetate, $CaNa_2$ EDTA - commonly used in cases of lead poisoning, also used to chelate zinc, copper, cadmium, chromium, manganese, and nickel
- Calcium edetate, $CaNa_3$ DTPA - used for transuranics such as plutonium and americium

Diuretics

Diuretics increase urinary excretion of sodium and water. Diuretics are used, on a limited basis, to reduce internal exposure; however, applications could include H-3, potassium-42 (K-42), chloride-38 (C1-38) and others. Diuretics can lead to dehydration and other complications if not performed properly.

Summary of Dose Reduction Methods

Nuclide	Medication	Comments
Am-241	DTPA by IV	Works on bone
Cf-252	DTPA by IV	
Cs-137	Prussian Blue	
Co-60	Stomach pump	
H-3	Increase H_20	
I-131	KI or KIO_3	Give within 2 hours
P-32	Phosphates	Isotopic dilution
Pu-239	Ca DTPA	Give within 2 hours
Sr-90	Sr or Ca by IV	

Process for Evaluating Bioassay Results

Doses should be based on radionuclide intakes, and biokinetic models should be used to interpret bioassay data and assess initial radionuclide intake. Methods for evaluating the various doses from intakes should be specified in the internal dosimetry technical basis document. The methods should be based on recommendations given in International Commission on Radiological Protection (ICRP) Publications 30, 48, and 54 and other reports of the ICRP and National Council on Radiation Protection and Measurements (NCRP) which embody improvements and updates of the science of internal dosimetry.

ICRP Publication 30 *Limits for Intakes of Radionuclides by Workers* has been used by DOE as the basis for its ALI and DAC listed in Appendix A of 10 CFR 835. Likewise, the modeling in ICRP 30 serves as the basis for interpreting the bioassay measurements in NUREG/CR4884, *Interpretation of Bioassay Measurements*. NUREG/CR-4884 uses the following equation to calculate intakes from bioassay measurements.

$$I = \frac{A(t)}{IRF(t)}$$

where: I = Estimate of intake

A(t) = The value of the bioassay measurement obtained at time t

IRF(t) = Intake retention fraction corresponding to the type of measurement for time t after estimate time of intake

ICRP 30 and 54 are based on general considerations (i.e., standard chemical forms and Reference Man metabolic modeling). Each individual's physiological characteristics and biochemical processes may be different. Periodic follow up bioassays can help determine the excretion rate of the radionuclide from the individual. This data can be useful in calculating the exact dose to the individual based on their metabolism, if necessary. Also, individual and facility specific factors should be used when more appropriate parameters are expected to change the dose calculations by more than a factor of 1.5.

The particulars of the exposure situation, such as particle-size distribution, will affect the lung compartment deposition fractions and the resultant biological clearances. For example, particles larger than 20 µm Activity Median Aerodynamic Diameter (AMAD) will deposit mainly in the nasopharyngeal (NP) region and tend to show biological retention and excretion characteristics more typical of an ingestion intake than an inhalation intake.

These characteristics are due to the fact that a large fraction of particles deposited in the NP region are cleared by the ciliated epithelial cells to the throat and subsequently swallowed. Fitting an individual's bioassay measurement data for a particular exposure situation to the standard modeling will, however, provide reasonably accurate estimates for most situations.

The following are important considerations for evaluating bioassay measurements:

- Determining the time of exposure. Accurate estimations of intake from bioassay measurements is dependent upon knowledge of time of intake.

- Excretion of radionuclides may be influenced by the workers diet, health condition, age, level of physical and metabolic activity, or physiological characteristics.
- Excretion of inhaled radionuclides from the lung depend on the particle size distribution.
- Appropriate measurement technique (in vivo or in vitro) based on the radionuclide decay characteristics.

- The effects of diuretics or chelation to reduce systemic uptake and to increase excretion rates.
- Representativeness of measurements such as 24-hour or accumulated urine or fecal measurements.
- The appropriate lung clearance class (D,W,Y).
- Particle size distribution.
- Chemical toxicity, as in the case of uranium.

Converting Radioactive Intake to Internal Dose

A way to calculate the CEDE to an individual is to use the tables of exposure to dose conversion factors for inhalation or ingestion found in *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report No. 11 (EPA 520/1-88-020). The CEDE to an individual is calculated:

CEDE = (Intake) (Dose Conversion Factor)

Another way to calculate the CEDE to an individual is to use a ratio of the intake to the stochastic ALI (sALI). The sALI for a radionuclide is that amount of the nuclide, which, if taken into the body by the specified route (inhalation or ingestion), would result in a CEDE of 5 rem. Thus, the CEDE may be calculated:

$$CEDE = \frac{I}{sALI} \times 5\ rem$$

where:

I = intake

A way to calculate a CDE to an organ or tissue is to use the tables of exposure to dose conversion factors for inhalation or ingestion found in EPA Federal Guidance Report No. 11. The CDE to an individual organ or tissue is calculated:

CDE = (Intake) (Dose Conversion Factor)

Another way to calculate the CDE to an organ or tissue receiving the largest dose is to use a ratio of the intake to the nonstochastic ALI (nALI). The nALI for a radionuclide is that amount of the nuclide, which if taken into the body by the specified route (inhalation or ingestion), would result in a CDE of 50 rem to an individual organ or tissue. Thus, the CDE to the organ or tissue for which the nALI is specified may be calculated:

$$CDE = \frac{I}{nALI} \; x \; 50 \; rem$$

where:

I = intake

For a detailed discussion of converting radioactive intake to internal dose, see Radiation Protection Competency 2.8.

Operation of External Dosimetry

Ideally, the instrument for recording dose should accurately and completely measure the amount of radiation absorbed by all tissues of the body. It should measure only the radiation of interest and be unaffected by moisture and temperature. The reading should not fade while the dosimeter is stored. It should cover a wide dose range and offer a high degree of accuracy over the entire range; results should be reproducible and easy to obtain. The dosimeter should be inexpensive and easy to wear. Unfortunately, the complexities of radiation interaction in different body tissues, limitations of dosimeter materials, and countless numbers of specific work situations makes the construction of such a device quite impossible.

Radiation type and energy, dose rate, exposure geometry, and tissue composition are all variables that must be identified in order to establish an individual radiation dose. Radiation rate surveys can be used to identify the general radiological environment such as radiation type and energy, exposure rate, etc. Individual dosimeters are issued to personnel in a radiation area to obtain the best estimate of each specific radiation dose. It is critical that the dosimeter be worn properly to accomplish this.

Dosimeters must be worn in such a position that the radiation exposure received by the dosimeter material accurately represents the radiation exposure received by the body. The dosimeter is, therefore, placed so as to be representative of the body mass as a whole (between the waist and shoulders on the front of the body). The measurement obtained is accordingly called the whole-body dose. Care must be taken not to shield the badge with materials that do not cover the entire body.

Additional dosimetry is required when the possibility exists for certain parts of the body to be more heavily irradiated than would be indicated by the whole body reading. There are specific recommendations for personnel dosimetry found in the *Radiological Control Manual*, Chapter 5, Part 1. Personnel who are expected to receive an annual external whole body dose greater than 100 mrem or an annual dose to the extremities greater than 10 percent of the limits specified in Table 2-1 of the *Radiological Control Manual* are to be provided personnel dosimetry at Department of Energy sites.

Handling radioactive materials directly with the hands and working directly with, or in close proximity to, sources of high-radiation are just a few of many instances where extra dosimeters may be needed. These dosimeters are worn to measure the highest dose received by an individual part of the body. The most common of these is a finger ring, incorporating a thermoluminescent dosimeter (TLD) chip in the flat part of the band. The flat portion of the ring must be worn facing either to the inside (palm) or outside of the hand, whichever orientation will measure the greatest dose received.

Accurate radiation dose measurement is a very important aspect of working in and around sources of radiation, and is essential to the maintenance of a radiologically-safe work environment. It is every worker's responsibility to ensure that their dose is accurately measured. With a conscious effort and a little care, this can be easily accomplished.

Photographic Film Dosimetry

In 1895, Wilhelm Roentgen, a German physicist, discovered darkening of photographic film in the room where he was experimenting with electrical discharges in vacuum tubes. Days later, he used these unknown rays, x-rays, to produce an image of his wife's hand on similar photographic film. Photographic film emulsions have now come to be widely used for the detection and measurement of various radiation types, most notably gamma, x-ray, and beta radiations. Radiation causes the film to darken and by measuring the degree of darkening, the radiation dose absorbed by the film is then determined.

The film badge is composed of one or more dental, small-sized, sheets of photographic film, wrapped in light proof wrappers. This film is housed in a plastic case which contains small pieces of three different metals. The thickness of the plastic case is selected in order that incident gamma radiation will produce electrons from interactions in the plastic, which then interact with the film to produce a darkening or image. The number of electrons emitted from the plastic must be equal to the number of electrons that subsequently leave the film. This process is termed electronic or charged particle equilibrium, and is required for the reading to be meaningful. When this condition is satisfied, an accurate representation of the energy that the radiation would deposit in nearby tissue is achieved.

Since gamma radiation may have energies from a few thousand to several million electron volts, metal foils of aluminum, copper, cadmium, and/or lead are placed in the film badge. These metals act as filters; that is, they provide varying amounts of shielding for the attenuation of different energies of gamma radiation. By comparing the exposure under the different filters, it is possible to determine an approximate spectrum of the gamma energies to which an individual was subjected.

All but the weakest beta radiation is capable of penetrating the paper covering the film. However, the majority of beta radiations will not penetrate the plastic case of the film badge. For this reason, a window is left open in the film holder to permit a determination of the beta exposure. The sensitive part of the film in these badges is a gelatin-base emulsion of silver halide grains, usually silver bromide. When radiation causes ionization in the film, some of the positive silver ions are transformed into electrically neutral atoms of silver metal. Upon developing, dark silver grains appear where the ionization has taken place. The degree of darkening is proportional to the radiation dose received by the film. The sensitivity of the film used in the badges is nearly independent of the gamma energy above 0.3 million electron volts. Below that figure, the darkening is highly energy-dependent, largely because the low-energy photons are attenuated by the emulsion in greater proportion than their roentgen value in air. Neutron-sensitive films are available, in which (n,p) reactions take place in the emulsion, resulting in proton recoil tracks on the film. The neutron dose is determined by computing the density of the tracks, often by visual counting under a microscope.

Processing of film includes developing, rinsing, fixing, washing, and drying. Detailed procedures vary with the type of solutions and film used, but specific procedures must be controlled within close limits to assure

accurate film densities. Both control film and exposed film standards are processed with each group of personnel film to assure that background radiation densities and calibration standard densities are correct after the film processing procedure.

Films used to measure beta and gamma radiation have dose ranges from 10 millirem to 1,800 rad for gamma radiation and 50 millirem to 1,000 rad for beta particles having a maximum energy greater than 400 keV.

Thermoluninescent Dosimetry

Radiothermoluminescence of fluoride (natural calcium fluoride [CaF]) was first studied by Wiedeman around 1903 while conducting his experiments dealing with the thermal conduction properties of various metals. During the period between 1924 to 1936, there were many experiments conducted and reported on by Prezibram. Their work was continued by numerous research scientists with little success. Thermoluminescent research and its application to the measurement of ionizing radiation began with the studies of Dr. Farrington Daniels at the University of Wisconsin in 1947, using the "pure" lithium fluoride (LiF) which was available at that time. The work was stopped for a few years because of poor results. In 1953, Dr. James H. Schulman of the U.S. Naval Research Laboratory developed a synthetic CaF dosimeter. In 1960, Dr. John R. Cameron reactivated thermoluminescent research at the University of Wisconsin under the guidance of Dr. Daniels. However, it was found that the LiF crystals had been purified to such an extent that they were no longer good for thermoluminescent research. A good grade thermoluminescent phosphor, called LiF (TLD 100) was developed at the Harshaw Chemical Co. The major impurity doped into the LiF was magnesium (LiF:Mg), but titanium (LiF:Ti) was also found to be important for high thermoluminescent sensitivity. In 1961, ConRad introduced the first commercially available thermoluminescent dosimeter reader (TLD Reader) and EG&G came out with a CaF and LiF TLD reader. At approximately the same time, MBLE, (manufacture Belge de Lampes et de Materiel Electronique, S.A.) a Belgian firm, released their military design of a TLD reader designed to read only natural CaF. Research has progressed to the point where today the use of TLDs for both personnel monitoring and environmental monitoring is the preferred method of dosimetry in the nuclear power field, and the Department of Energy.

Thermoluminescent dosimeters employ crystals that come under the category of scintillation detectors. The crystal lattice of the thermoluminescent phosphor contains conduction bands, which form the basis for the entrapment of energetic electrons coming from ionized phosphor atoms.

Thermoluminescence dosimetry consists of two steps. The first is radiation exposure that leaves some excited electrons in metastable states; the second is read-out, during which the exposed TLD is heated and the resulting emission of light measured.

Thermoluminescent phosphor crystals contain valence and conduction bands, just as the scintillation and semiconductor crystals previously discussed. The electrons that are freed from the valence band, as a result of the radiation interacting with the phosphor, are raised to the conduction band. They move around freely until they become trapped in a metastable state. The metastable state is associated with defects in the crystal lattice structure caused by the presence of impurities. The trapped electrons, in this case, are prevented from spontaneously going back to the valence band. The electron can be held in the metastable state for a considerable length of time.

The light emissions, which are measured to determine radiation dose, will not occur unless the trapped electrons can somehow recombine with a hole and give up its excess energy. The electrons must be freed from the metastable state to begin the recombination process that will ultimately result in the emission of thermoluminescent photons. The electrons are freed when heat is applied to the phosphor. This provides the electrons with sufficient energy to raise back into the conduction band and be free to move around until they recombines with a trapped hole.

The amount of energy required to free an electron from a metastable state is a function of the energy gap between the valence and conduction bands. Different thermoluminescent materials have different band gaps and thus require different amounts of applied heat to free all the trapped electrons. In a single crystal material, the probability of releasing a trapped electron increases as the temperature of the applied heat is increased. Some trapped electrons are released at low temperatures and others require the maximum heat cycle temperature. The intensity of emitted light, because it is a function of the number of electrons recombining, will be weak at low temperatures, pass through a maximum intensity at high temperatures, and then fall to zero when there are no more trapped electrons.

The most effective thermoluminescent materials have strong light outputs and are able to maintain the electrons in metastable states at normal temperatures.

A large proportion of the TLDs in use are LiF crystals. Their popularity is derived from their excellent storage stability and air/tissue equivalent density. Dose response is reasonably constant for gamma energies from 10 keV to 10 MeV. Dose response is linear from 0.01 rad (100 µGy) to approximately 1,000 rad (10 Gy). However new materials with higher sensitivities and linearity at high doses (> 1,000 rad) are replacing LiF.

Some other compounds using Li are given below with their applications:

- The $Li_2B_4O_7$:Cu element is designed to evaluate skin dose equivalent values.
- The $Li_2B_4O_7$:Cu has a plastic shield for reading deep dose equivalent values.
- The $CaSO_4$:Tm elements are designed for low doses, enabling daily or arbitrary checks of doses.

The energy response of thermoluminescent dosimeter material varies with many factors for each specific phosphor. However, TLD energy response is generally greater than photographic film but not as good as radiophotoluminescent (RPL) glass dosimetry. Finer grain phosphors generally exhibit better radiation sensitivity than those of coarser grain. The sensitivity of any inorganic TLD is increased with the addition of hydrogenous material.

TLD Reading

TLD readers come in a variety of models. Some are designed to read TLD powder or chips and others read TLD materials on cards. Some instruments are manually operated, that is, only one TLD at a time can be read. This method of operation applies to those instruments that read TLD powder or chips. Automated instruments can be loaded with many TLD cards so that the complete operation from reading to printout is done automatically.

The heating cycle is a very important aspect of an instrument's ability to properly measure the dose recorded by the TLD. The preheat portion of the cycle removes all electrons from low temperature traps. Light emitted as these electrons recombine is not measured by the reader. Following preheating, the temperature within the reader increases to the readout temperature of the heating cycle, normally about 285°C. During the readout cycle, the light output from high temperature traps, as shown by the glow curve, is measured, interpreted, and the total absorbed dose (mR, R, kR) is displayed on the front of the instrument when the cycle is complete.

Comparison of Thermoluminescent and Film Dosimeters

TLDs offer many advantages over film badges and, in fact, are replacing film badges in many instances. Well over 2,000 natural minerals exhibit thermoluminescence properties allowing for a wide range of material selection depending on the particular need at hand. TLDs are capable of a broad exposure range (µR to 10⁵ R) and are less energy dependent, particularly at low gamma energies, than photographic film. Unlike film, they are reusable and are exempt from the chemical variability and inconvenience experienced with time-consuming, expensive photographic development. They also offer good storage stability.

A major disadvantage of the TLD, however, is that the TLD offers no permanent dose record because the radiation dose is released from the crystal as it is read. If a mistake is made during reading, there is no way to recheck the results. This problem can be remedied by using a graphical ink pen plotter to record each glow curve or TLD reader with printing capacity. Fading, which is the amount of signal loss over time, is also an important factor in determining the type of dosimeter. For TLD badges, the higher the temperature required to trap an electron, the less the fading will occur. Photographic film can be kept indefinitely with a permanent image, however, film badges are rarely utilized at DOE facilities.

Albedo Dosimeters

TLDs used to detect neutrons incorporate two isotopes of lithium, Li-6 and Li-7, both of which are equally sensitive to gamma radiation. However, Li-6 has a large cross section for the thermal neutron (n,α) reaction. Production of the alpha particle initiates the thermoluminescence process that ultimately results in a measure of the dose due to thermal neutrons; whereas, Li-7 is relatively insensitive to thermal neutrons. The Li-6 phosphor will read both neutron and gamma radiation interactions; whereas, the Li-7 phosphor will read only gamma interactions. Neutron dose is determined by subtracting the Li-7 reading (γ) from the Li-6 reading $(n+\gamma)$ and applying a conversion factor to the difference.

The term albedo stands for reflecting. Some of the thermal neutrons detected by the Li-6 are originally fast neutrons that interact with hydrogen in the body, are thermalized, reflected or scattered off the body and detected. This makes the albedo dosimeter position sensitive; therefore, it must be properly orientated. Because the neutrons can be moderated to thermal energies, they are reflected from the body through the back of the badge into the albedo dosimeter. Therefore, it is important to wear the dosimeter extremely close to the body (on the flesh) to obtain accurate measurements. The front of the badge is shielded with cadmium to reject external thermal neutrons.

Personnel Criticality Dosimeters

The determination of dose from a criticality accident, which emits a spectrum of neutron energies, can be accomplished by the use of foil activation. One popular type of criticality dosimeter uses indium, cadmium-covered indium, gold, cadmium-covered gold, sulfur, and cadmium-covered copper. By measuring the neutron induced activity of these foils, a breakdown of the neutron energies and intensity of the neutron flux gives an estimate of the neutron dose.

Pocket Dosimeters

Pocket dosimeters are compact, easy-to-carry devices that indicate an individual's accumulated dose of radiation at anytime, thus eliminating the delay of film badge/TLD processing. However, because of the possibility of faulty readings, due to rough treatment, the dosimeter reading does not constitute a permanent legal record of dose received. A pocket dosimeter can be self-reading or not. In the self-reading type, a small compound microscope is used to observe the response. The type which is not self-reading, called the pocket chamber, is similar in construction to the self-reading type, but another instrument called the charger reader must be used to read it. The self-reading type is normally preferred since it can be read anywhere and at any time.

A self-reading pocket dosimeter (SRPD) consists of a small air-filled chamber in which a quartz fiber electrometer, a small microscope and a graduated scope across which the shadow of the quartz fiber moves to indicate the applied dose, is suspended.

The design and operation of a self-reading pocket dosimeter utilizes the principle of discharging a pair of opposite charged surfaces when the air between them is exposed to ionizing radiation. The electric charge required to attract the ionized gas particles is impressed on the electrometer and the chamber wall by means of a suitable charging unit. Ionizing radiation penetrating the chamber forms positively and negatively charged gas particles. These charged particles are attracted to the oppositely charged surface; i.e., the negative particles to the electrometer and the positive particlesto the chamber wall. The migration of the negative particles to the electrometer permit the fiber to move closer to the frame which, in turn, causes the shadow of the fiber to move across the calibrated scale.

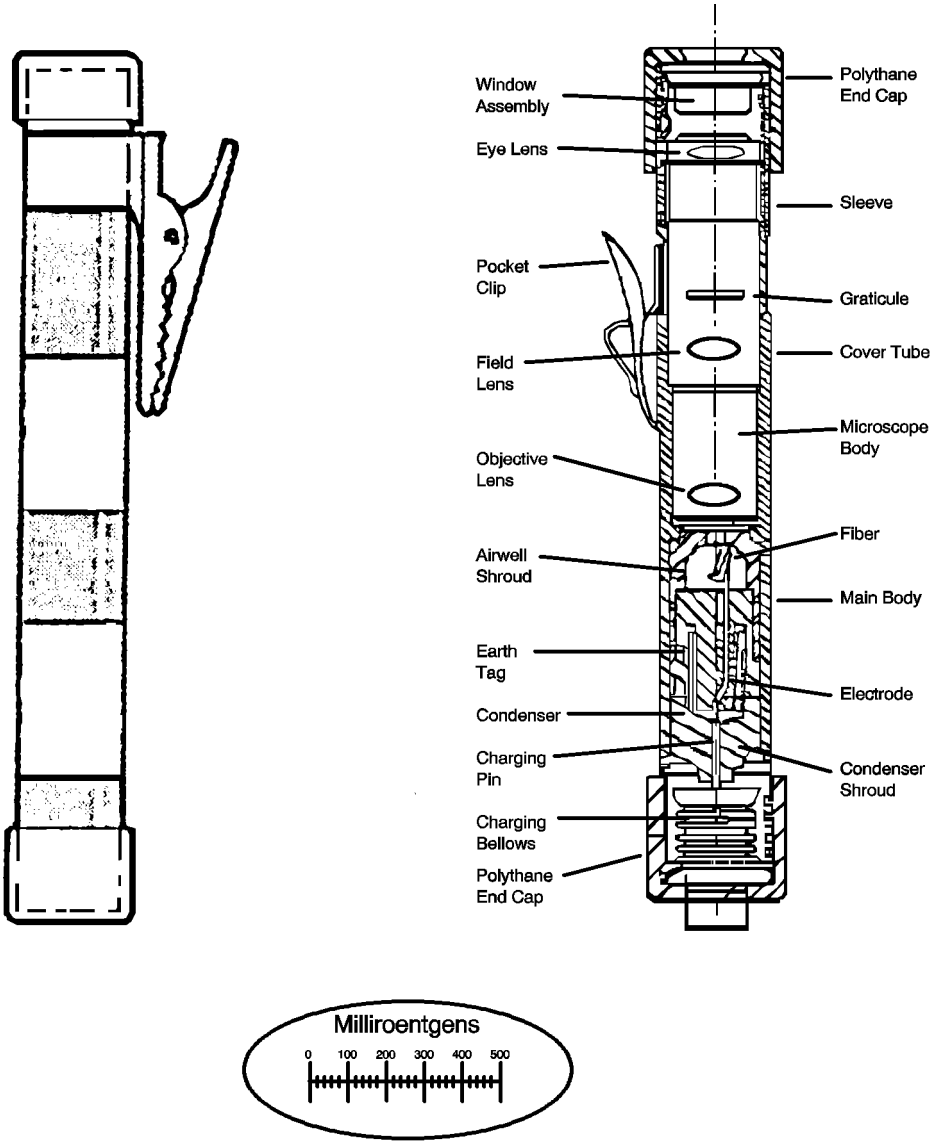


Figure 1 - SRPD

Pocket dosimeters are available in many ranges of gamma exposures from 0 through 200 milliroentgens (mR) to 0 through 1,000 roentgens (R). The sensitivity of the instrument is determined at the time of manufacture by the selection of the electrical condenser incorporated in the unit.

The dosimeter charger is a small, portable, battery-operated power supply designed to impress a DC voltage on the charging electrode of a dosimeter. In use, the dosimeter is inserted into the charging well and pressed down firmly. Pressing the dosimeter into the well serves two purposes: (1) it actuates a switch that turns on a bulb so the fiber of the electrometer may be seen, and (2) it closes the charging switch in the end of the dosimeter so that an electrical connection is made between the charging circuit and the electrode. While holding the dosimeter firmly in the well, an adjusting knob is rotated until the hairline is on a mark just below zero. Frequently, a capacitance effect causes the hairline to shift slightly when releasing the contact with the charging well. This may necessitate several charging attempts, setting the hairline slightly above or below the mark, to accomplish a zero setting when the dosimeter is removed from the charging well. However, it is not necessary to have an exact zero setting since the exposure received is determined by subtracting the initial reading from the final reading.

Pocket dosimeters should be worn on the upper front part of the body close to the dosimeter of record (e.g., the TLD). Pocket dosimeters are quite rugged and durable; however, they are designed to be worn on a person's clothing and should not be subjected to any greater shock or abuse than one would expect during normal active work.

Electronic Dosimeters

Electronic dosimeters are replacing the self-reading pocket dosimeter as the secondary dosimeter at nuclear plants in the United States. Electronic dosimeters, called EDs for short, provide a more reliable, user-friendly device to assist workers and plant staffs in reducing radiation dose. Most EDs on the market provide the following features:

- Audible alarms for accumulated dose and dose rate.
- Easy to read displays of accumulated dose and dose rate.
- Durable readings that are not affected by physical activity (drift).
- The ability to monitor a wide range of accumulated dose and dose rate.

Advanced models can provide the following additional features:

- Real time dose monitoring using radiotelemetry.
- Storing and manipulating radiation dose data for ALARA programs, dose records, and radiation work permit (RWP) generation.

ICRP Dose Concepts

Since almost every exposure of the body involves the irradiation of more than one tissue, it is appropriate to recommend a dose equivalent limit based upon the total risk of all tissues irradiated. The ICRP 26 dose limitation system sets a single dose equivalent limit for uniform irradiation of the whole body and implements a subsystem designed to ensure that the total risk from irradiations of parts of the body does not exceed the risk from uniform irradiation of the whole body.

For occupational exposures, it is appropriate to assess the levels of risk associated with the dose equivalent limits. A valid method for judging the acceptability of the level of risk in radiation work is to compare this risk with the risk of other occupations recognized as having high standards of safety. Safe industries are considered to have an accident rate of less than or equal to 10^4 fatalities per year. Toward that end, the ICRP 26/30 recommends the following limits which are based on risk.

ICRP 26/30 Recommended Dose Equivalent Limits

Limit to prevent non-stochastic effects to all tissues except the lens of eye	0.5 Sv (50 rem)
Limit to prevent non-stochastic effects to the lens of eye	0.3 Sv (30 rem)
Limit to limit stochastic effects to an acceptable of risk	$\sum_T w_T H_T \leq 50\ mSv\ (5\ rem)$ where: w_T = the tissue weighting factor representing the proportion of stochastic risk resulting from irradiation of tissue (*T*) to the total risk when the whole body is irradiated uniformly H_T = annual dose equivalent in tissue (*T*)

One of the basic concepts is that the total risk of radiation exposure should include the risks from both external and internal exposures. This is accomplished bydeveloping a means to sum external and internal doses together and limit the total dose to 5 rem per year. The ICRP in report 26/30 developed the concept of total effective dose equivalent or TEDE for short. The TEDE is calculated by adding the internal dose, expressed as the committed effective dose equivalent (CEDE), to the external dose, expressed as the deep dose equivalent (DDE) obtained from external dosimetry data at a measurement depth of 1 cm in tissue (a density thickness of 1,000 mg/cm^2). This calculation may be represented as follows:

$$TEDE = DDE + CEDE$$

External Dose Determination

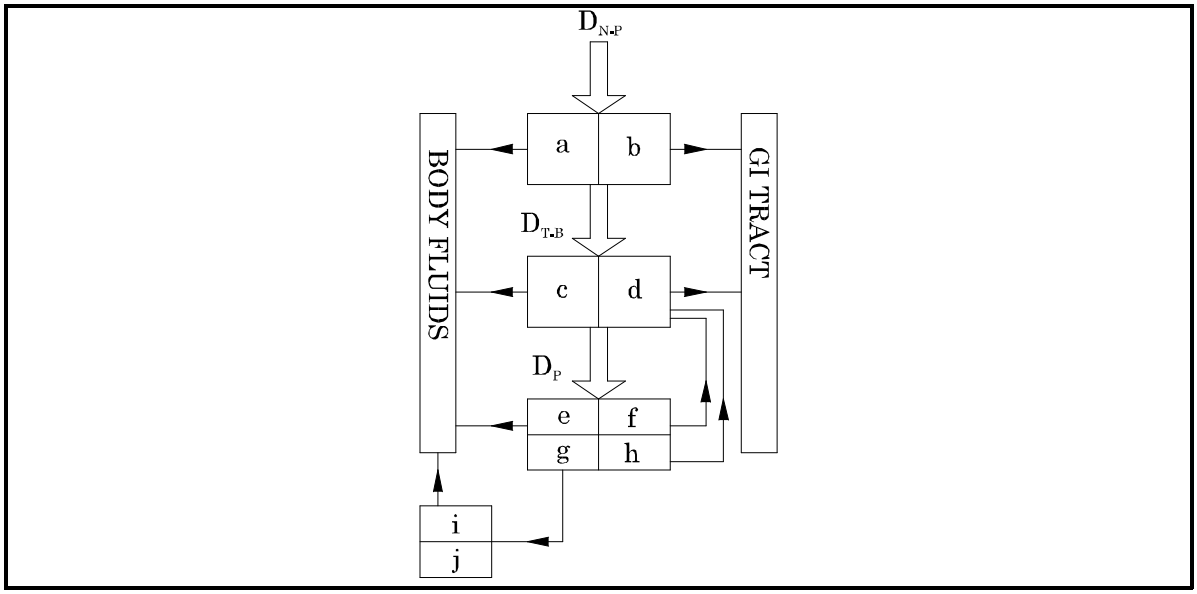
External dose is determined by monitoring individuals with dosimetry devices that are worn at all times in radiation areas. External limits are specified based on the depth in tissue that the radiation is capable of penetrating. The following chart lists the various tissue depths and limits.

Dose Equivalent Type	Abbreviation	Measurement Depth for External Sources (cm)	Annual Limit (rem)
Deep dose equivalent	DDE	1	5
Lens of eye dose equivalent	LDE	0.3	15
Shallow dose equivalent, skin or any extremity	SDE, WB	0.007	50

Internal Dose Determination

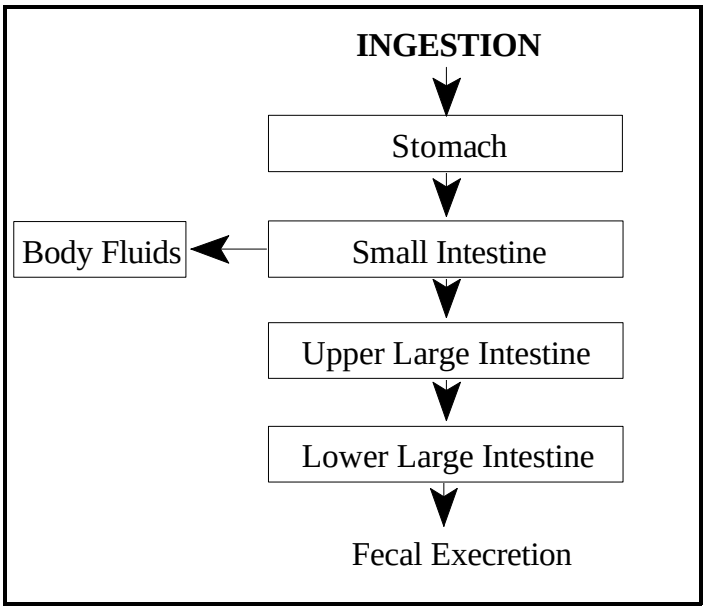
Calculating internal dose is much harder than calculating external dose. Because of variations among individuals' size and metabolism, it has been found advantageous to define the physical and chemical properties of a representative individual. Toprovide a standard for evaluating the effects of radiation on the body, averages are determined for the physical and chemical makeup of the body. This average is called Reference Man. (See attachment A for the specific values of Reference Man.) Two of the more important models developed for Reference Man are the respiratory tract and the gastrointestinal tract.

Respiratory Tract Model



The respiratory tract model is heavily dependent on the particle size of the radioactive material breathed in. The above model assumes a particle size of 1 micron.

Gastrointestinal Tract Model



Using the Reference Man metabolic retention times and dimensions, calculations were made to determine the amount of radioactive material taken into the body by the specified route that satisfies the following dose limit conditions:

$$\left(\sum_{T} \; w_T \; H_T \; \leq \; 50 \; mSv \; [5 \; rem] \right)$$

as well as non-stochastic effects:

$$(H_T \leq 0.5 \text{ Sv } [50 \text{ rem}]).$$

The amount of radioactive material taken into the body is called the ALI. The ALI is the quantity of radioactive material which, if inhaled or ingested in one year, would irradiate a person and result in a stochastic dose of 5 rem or a nonstochastic dose of 50 rem, whichever is less.

The DAC is calculated using the ALI and is the concentration of radioactive material in air that, when breathed by Reference Man for 2,000 hours in a year under conditions of light activity, would result in an intake equal to the ALI. The DAC calculated from the following formulas depending on whether the limit is stochastic or nonstochastic:

$$STOCHASTIC \; DAC \; = \; \frac{sALI}{2{,}400 \; m^3}$$

$$NONSTOCHASTIC \; DAC \; = \; \frac{nALI}{2{,}400 \; m^3}$$

Committed Dose System

The rate that radioactive materials are excreted from the body varies significantly. Because of this variation, the dose delivered via internal depositions are integrated over time. What this means is an intake that occurs today may continue to deliver a dose to the individual for many years. A time period must be considered in order to assign a dose to the individual. In ICRP 26/30, this time period is 50 years. From any intake, the integrated dose is calculated for a 50-year period and the total dose is assigned in the year that the intake occurred.

A logical order in which to calculate dose equivalents, to demonstrate compliance with the annual limits, is to begin with internal dose equivalents to organs or tissues. Once the CDE are known for all organs or tissues, the CEDE may be calculated. In order to relate risk of the dose to an organ to the risk of the whole body, apply tissue weighting factors (w_T) to the CDE. The w_T are given in the definitions section of this lesson. There are two ways to calculate the CEDE. One way is to multiply the CDE by the weighting factor, and the second way is to divide the intake by the sALI.

The CEDE and CDE are then added to the DDE obtained from external dosimetry to demonstrate compliance with the annual limit of 5 rem TEDE.

Estimation of Dose

Estimates of both internal and external doses can be made utilizing surveys, measurements, and calculations.

Methods for External Dose Estimates

If external dosimetry is lost, the individual should safely stop work and leave the area. Determining the dose to the individual can be accomplished by surveys, measurements, cross references, and calculations.

In the case of lost dosimetry, the first step would be to check if the individual has a secondary dosimeter such as a pocket or electronic dosimeter. In most cases the secondary dosimeter can provide a reliable estimate of worker exposure. If both primary and secondary dosimetry have been lost, dose estimates can be obtained from other workers from the same radiation field. Another means of dose estimation would be to dispatch health physics personnel to survey the area, document radiation levels, and calculate the dose based on stay time. Area radiation monitors may also provide dose rate data, if one is nearby. Estimations of dose rates can be made based on the activity of radioactive material suspected to be present. Previous surveys of the area such as the RWP surveys or records of previous similar jobs can be used, if necessary. Interviews with workers can be conducted by the health physics staff to obtain exact details of the incident. Only in extreme cases would a reconstruction of the event, covering the exact place the individual was standing and time spent in the area, need to take place.

Methods for Internal Dose Estimates

Internal dose estimates are based on the intake, or the amount, of radioactive material taken into the body. Intakes can be estimated by either bioassay or calculations from airborne concentrations and stay times. Internal dose estimates should be based on bioassay data rather than airborne concentrations, unless the bioassay data is:

- Unavailable.
- Inadequate.
- Demonstrated to be less accurate when compared to airborne concentrations.

When available bioassay data is lacking or contradictory, professional judgement will be needed to make a dose evaluation. Since the evaluations of internal dose depend on knowing the intake profile with respect to time, the dose evaluation staff should base the time period of the intake on known incidents, air monitoring data, records of perturbations in facility operation, and/or discussions with the workers by the radiation protection staff. If the time course of intake cannot be plausibly established, the procedure for evaluating doses based on the internal dosimetry technical basis document should be used.

Airborne Dose Calculations

ALI and DAC values are both radionuclide specific. The DAC values can be obtained from Appendix A of 10 CFR 835, listed in both µCi/ml and Bq/m^3. The DAC fraction, or percent of the DAC, is calculated by the following formula:

$$DAC\ Fraction = \frac{measured\ airborne\ concentration}{appendix\ A\ DAC}$$

The airborne concentration and DAC limit should be in the same units and, therefore, the DAC fraction is a unitless ratio. If more than one radionuclide is present in the sample, then the DAC fraction is calculated using the sum of the fractions rule:

$$DAC\ Fraction = \frac{Concentration\ A}{DAC\ A} + \frac{Concentration\ B}{DAC\ B} + \frac{Concentration\ C}{DAC\ C}$$

The stay time is necessary to calculate the dose from airborne radioactivity. The DAC fraction, calculated above, is multiplied by the stay time in hours and the product is called DAC-hours (DAC-hrs). The DAC-hrs are divided by any respiratory protection devices and multiplied by a dose conversion of 2.5 mrem per DAC-

$$Dose = \frac{airborne\ concentration}{appendix\ A}\ \frac{stay\ time\ (hrs)}{1}\ \frac{1}{respiratory\ protection\ factor}\ \frac{2.5\ mrem}{1\ DAC-hr}$$

hr for a stochastic DAC value or 25 mrem for a nonstochastic DAC value. The following formula represents the dose calculation for a stochastic DAC. The respiratory protection factor is defined as the ambient airborne concentration divided by the concentration inhaled (for detailed information on respirator use and regulations see *Practices for Respiratory Protection*, American National Standards Institute, ANSI Z88.2-1992).

ATTACHMENT A

Specifications for Standard Man

	Adult Man	Adult Woman
Weight (kg)	70	58
Length (cm)	170	160
Surface Area (cm^2)	18,000	16,000
Specific Gravity	1.07	1.04
Total Body Water (mL/kgW)	600	500
Extracellular Water	260	200
Intracellular	340	300
Total Blood Volume (mL)	5,200	3,900
Red Cell Volume (mL)	2,200	1,350
Plasma Volume (mL)	3,050	2,500
Total Blood Weight (g)	5,500	4,100
Red Cell Weight (g)	2,400	1,500
Plasma Weight (g)	3,100	2,600
Total Adipose Tissue (kg)	15	19
Subcutaneous	7.5	13
Sparable	5.0	4
Yellow Marrow	1.7	1.4
Interstitial	0.8	0.6
Total Connective Tissue (g)	5,100	4,100
Cartilage	2,500	2,000
Tendons and Fascia	850	750
Other	700	1,400

ATTACHMENT A

Specifications for Standard Man (Cont'd)

	Adult Man	Adult Woman
Total Fat (kg)	13.5	15
Nonessential	12	13.8
Essential	1.5	1.2
Hair (g)	20	300
Nails (g)	3	3
Skeletal Muscle (kg)	28	17
Total Skin (g)	4,900	3,500
Epidermis	500	400
Dermis	4,400	3,100
Hypodermis	7,500	13,000
Resting Metabolic Rate (cal/min-kg)	17	16
Oxygen Inhaled	920	640
Carbon Dioxide Exhaled (g)	1,000	700
Total Lung Capacity (liters)	5.6	4.4
Functional Residual	2.2	1.8
Vital	4.3	3.3
Dead Space	0.160	0.130

3. SELF STUDY SCENARIOS/ACTIVITIES AND SOLUTIONS

Scenario 1

On May 31, two workers employed at a DOE contractor facility were tasked with installing a new process line in an indoor building posted and controlled as a high-radiation and high-contamination area. This activity was infrequently performed. Both workers had completed Radiological Worker I training and additional training to allow them access into high-radiation areas. Both workers were currently in compliance with 10 CFR 835 training requirements. However, one worker (Worker "A") required retraining effective the first day of the following month. The workers had been issued and had signed a Radiation Work Permit (RWP) limiting the scope of work to installing the new line in a shielded area of the building. The RWP required a full set of protective clothing without respiratory protection based on the scope and location of the work. Personnel dosimetry requirements consisted of a pocket ionization chamber (0 to 200 mR scale) and a thermoluminescent dosimeter (TLD) badge.

The workers entered the area and began installng new pipe. Operations continued smoothly until late in the afternoon when the workers discovered an out-of-service drain line interfering with installation of the new line. Unfortunately, they failed to observe a faded "Caution: Radioactive Materials" posting placed on the drain line. Because of the time, they decided to quit for the day.

The following morning, the workers informed their supervisor of the situation. The supervisor determined that work could not continue until a flanged pipe tee, connected to the drain line, was removed. Worker "A" attempted to remove the pipe tee, but, having difficulty loosening it, asked for assistance from Worker "B". After five minutes and considerable effort, the tee was successfully removed. Worker "B" observed that one of his gloves had been badly torn during this process, so he removed it and left it on the floor. He then spent a couple of minutes closely examining, touching, and measuring the end of the drain in order to locate a cap that would fit the exposed opening. Not finding an appropriate match, he decided to leave the end open. The two workers spent the following ten minutes one foot away from the old drain line while connecting another section of the new process line. After installation was completed, the workers departed the work area, removed their protective clothing, and performed whole-body frisking. Worker "A" was free of contamination; Worker "B" found contamination on his hands. A radiological control technician (RCT) was notified.

Following decontamination of Worker "B's" hand, the RCT performed a survey near the drain line. His instrumentation indicated a whole-body dose equivalent rate of 60 mrem/hr at a distance of 30 centimeters. The RCT observed that the open end of the drain line contained an unknown residue. Taking adequate precautions, he collected samples from the drain line; isotopic analyzes performed immediately after collection revealed the presence of plutonium-238 (Pu-238) and plutonium-239 (Pu-239) in the nitrate form. Because of the potential and concern for internal deposition of radioactive material, urine and fecal samples from both workers were obtained for the next several days. Results for Worker "A" were negative. Bioassay results for Worker "B" indicated an intake of 10 Bq of Pu-238 and 12 Bq of Pu-239.

1. What are some concerns raised in this scenario?
2. Estimate the whole body external dose equivalent received by Workers "A" and "B" due to exposure from the out-of-service drain line only.

NOTE: To aid you in your calculation, assume that the workers maintained a constant one-foot distance from the drain line and:

- each worker initially spent five minutes at the drain line discussing what to do about the pipe tee obstruction interfering with their work.
- each worker spent five minutes attempting to remove the flanged pipe tee.
- Worker "B" spent an additional two minutes examining the exposed drain opening.
- each worker spent ten minutes next to the drain line connecting another section of the new process line.

The equation to calculate the external dose equivalent (H) is:

$$H = \textit{dose equivalent rate x time}$$

3. Use the information and equations provided below to calculate the committed effective dose equivalent (CEDE) to Worker "B" and the committed dose equivalent (CDE) from these intakes.

Table of Dose Conversion Factors (DCF)			
Radionuclide	**Class**	**CDE per Unit Intake (Sv/Bq)**	
------	------	Bone Surfaces	Effective
Pu-238	W	1.90E-3*	1.06E-4*
Pu-239	W	2.11E-3*	1.16E-4*

* Taken from EPA Federal Guidance Report No. 11, p. 151

The equation to calculate the CEDE is:

$$H_{50,E} = (Intake)(\Sigma DCF)$$

The equation to calculate the CDE to bone surfaces is:

$$H_{50,BS} = (Intake)(\Sigma DCF)$$

4. What is the significance of the reported doses in terms of DOE limits?

Scenario 2

Calculate the dose to an individual who works in anarea for 3 hours wearing a full face negative pressure respirator. The airborne concentrations are 5×10^{6} µCi/ml for Co-60 and 7×10^{8} µCi/ml of Cs-137

Given:

Stochastic DAC limit for Co-60 = 1×10^{8}

Stochastic DAC limit for Cs-137 = 7×10^{8}

Respiratory Protection = 50

Activity Solutions:

Scenario 1, Solution

1. DOE 10 CFR 835 and the *Radiological Control Manual* address some of the concerns in this part of the scenario.
   - The faded radiological posting present on the drain line is a concern. According to 10 CFR 835 Section 601, signs shall be "clear and conspicuously posted." Article 231 of the *Radiological Control Manual* states that postings should "alert personnel to the presence of radiation and radioactive materials," "be conspicuously posted and clearly worded," and "be maintained in a legible condition." The worker's failure to observe the posting is clearly not entirely their fault, but likely resulted in Worker "B" receiving a higher dose.
   - The reading of 60 mrem/hr at 30 cm qualifies as a radiation area under 10 CFR 835 Subpart A, Section 835.2 and as noted in Table 2-3 of the DOE *Radiological Control Manual*. Posting the drain line as a radiation area should have been performed under Article 234.
   - 10 CFR 835.402 requires monitoring in the workplace for exposures to internal radiation. Articles 136 and 361 from the *Radiological Control Manual* refer to the difficulty in measuring transuranic uptakes. For that reason, considerable attention should be paid to controlling and preventing internal exposures. Article 316 cites the need for appropriate engineering and administrative controls as primary and secondary methods, respectively, to limit internal exposures. Respiratory protection is the next resort. Because: (1) respiratory protection was not required on the RWP based on the original scope of work (no potential for airborne radioactivity was thought to exist), and (2) the significance in the change in job scope was not recognized by the workers or the work supervisor, respiratory protection was not utilized at the time the pipe obstruction was discovered, removed, and opened. As a result, one of the workers received an internal dose.
   - Annual allowable dose limits are provided in Subpart C, Section 202 of 10 CFR 835 and Article 213 of the *Radiological Control Manual*. While the whole body and organ limits were not exceeded in this case, the doses received by the workers were not maintained ALARA.

2. Calculating the external whole-body dose equivalent received by the workers can only be estimated in this case because there are uncertainties regarding: (1) general exposure rates in the shielded portion of the building where they were working (no information was provided), and (2) the workers' proximity to the drain line at any given time. A constant one-foot distance was chosen to simplify the calculation Given these uncertainties, the whole body doses are estimated as follows:

   <u>Worker "A"</u>

   Worker "A" spent an estimated 20 minutes near the drainline. Therefore, the worker received a dose equivalent of:

   (60 mrem/hr) *x* (1 hr/60 minutes) *x* 20 minutes = **20 mrem**

Worker "B"

Worker "B" spent an additional two minutes near the drain line. The dose equivalent is:

(60 mrem/hr) *x* (1 hr/60 minutes) *x* 22 minutes = **22 mrem**

3. The CEDE is calculated as follows:

$H_{50,E}$ = [(10 Bq x 1.06 E-4 Sv/Bq) + (12 Bq x 1.16 E-4)]

= 2.45 E-3 Sv

= **0.245 rem (245 mrem)**

The CDE is calculated as follows:

$H_{50,B}$ = [(10 Bq x 1.90 E-3 Sv/Bq) + (12 Bq x 2.11E-3 Sv/Bq)]

= 4.4 E-2 Sv

= **4.4 rem**

4. Considering only the dose received from exposure to the drain pipe, neither the DOE annual whole-body limit of 5 rem (from both internal and external radiation) nor the organ/tissue dose limit of 50 rem was exceeded for either worker. A summary of the doses received by both workers is summarized in the table below.

SUMMARY OF DOSES RECEIVED BY WORKERS			
Worker	**External Dose (mrem)**	**Internal Dose (mrem)**	**TEDE**
A	20	-----	20
B	22	245	267

Worker "A" received an external dose of 20 mrem and no internal dose for a total effective dose equivalent (TEDE) of 20 mrem, while Worker "B" received a TEDE of 267 mrem (245 mrem internal and 22 mrem external). No information was provided in the scenario as to whether any facility administrative control limits (ACLs) were exceeded. The ALARA philosophy suffered in this instance, however; therefore, an ALARA review should be initiated to prevent this situation in the future.

Scenario 2, Solution

Calculate the dose to an individual who works in anarea for 3 hours wearing a full face negative pressure respirator. The airborne concentrations are 5 x 10^{6} µCi/ml for Co-60 and 7 x 10^{8} µCi/ml of Cs-137

Given:

Stochastic DAC limit for Co-60 = 1 x 10^{8}

Stochastic DAC limit for Cs-137 = 7 x 10^{8}

Respiratory Protection = 50

$$Dose = \left(\frac{5x10^{-6}\ \mu Ci/ml}{1x10^{-8}\ \mu Ci/ml} + \frac{7x10^{-8}\ \mu Ci/ml}{7x10^{-8}\ \mu Ci/ml} \right) \frac{3\ hours}{PF\ 50} \frac{2.5\ mrem}{1\ DAC\ hour} = 75.15\ mrem$$

4. SUGGESTED ADDITIONAL READING AND/OR COURSES

Readings

- 10 CFR 835, *Occupational Radiation Protection.*
- DOE N 441.1, *Radiological Protection for DOE Activities.*
- DOE/EH-0256T (Revision 1), *Radiological Control Manual.*
- DOE Order 5480.4, *Environmental Protection, Safety, and Health Protection Standards.*
- G-10 CFR 835, Revision 1, *Implementation Guides for Use with Title 10 Code of Federal Regulations 835*.
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